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United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Attention: Mr. Patrick Sears
Senior Project Manager
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

References:
(a) License No. DPR-3 (Docket No. 50-29)
(b) Letter, NRC to Yankee Atomic Electric Company, dated August 31, 1990
(c) Letter, Yankee Atomic Electric Company to NRC, dated November 28, 1990
(d) Letter, Yankee Atomic Electric Company to NRC, dated November 26, 1990
(e) Letter, NRC to Yankee Atomic Electric Company, dated December 11, 1990

Subject: Reactor Pressure Vessel Plan

Dear Sir:

The NRC Safety Assessment of the Yankee reactor pressure vessel (Reference (b)) required Yankee to submit a plan to address the uncertainties contained in the Safety Assessment. Upon clarification from the NRC staff, the submittal was to be made 60 days from startup from the 1990 outage and not 60 days from August 21, 1990 as stated in Reference (b). The outage startup date was November 11, 1990.

The plan for addressing the uncertainties are attached. The plan covers:

1. Inspection methods for the beltline welds and each beltline plate to determine if the metal contains flaws (Attachment A).

2. Testing on typical Yankee vessel plate material to determine the effects of irradiation, austenitizing temperature, and nickel composition on embrittlement at 500°F and 550°F irradiation temperatures (Attachment B).

3. Sampling of the circumferential and axial welds in the beltline (Attachment C).

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4. Installing surveillance capsules in accelerated irradiation positions in the Yankee pressure vessel (Attachment D).

Reference (b) also requested an updated fluence calculation and peer reviews of Yankee's July 5, 1990 reactor vessel evaluation. The fluence calculation was submitted November 28, 1990 (Reference (c)). The peer reviews have been drafted and are expected to be submitted by the end of January.

If you should have any questions with the attached plan, please notify us.

Sincerely,

John D. Haseltine

John D. Haseltine
Director
Yankee Project

JDH/gjt/WPP77/241

Attachments

cc: B. Elliot (NRC, NRR)
R. Wessman (NRC, NRR)
W. Russell (NRC, NRR)
USNRC Region I
USNRC Resident Inspector, YNPS

ATTACHMENT A

REACTOR VESSEL INSPECTION

INTRODUCTION

Yankee presented its preliminary inspection program to the NRC staff during a meeting at the EPRI NDE Center on November 1, 1990. The discussion with the NRC included methods to implement the inspections, how they would be qualified, and the inspection schedule. The NRC Safety Assessment requires Yankee to demonstrate the methodology for inspection by the end of the present fuel cycle but does not require the performance of the inspection at the next outage. It is Yankee's plan to demonstrate and perform the inspection at the next outage. If for unforeseen reasons the inspection cannot be completed, then the schedule will allow us to use the following outage for a follow-up inspection.

OBJECTIVES

The inspection objectives are:

1. Perform an inspection of the vessel shell, in the beltline region, behind the thermal shield from the cladding surface to a depth of 1" into the base metal. Inspections shall be performed such that a 1/4" perpendicular planar flaw can be resolved at the base metal surface and through the inspection volume.

2. Develop a method to discriminate and size flaws detected in Item 1.

3. Qualify equipment, procedures, and personnel to ASME Section XI, Appendices 7 and 8, or to a qualification procedure which meets the intent of these appendices.

INSPECTION METHODS

The inspection methods for the Yankee vessel are controlled by the limited access behind the thermal shield, the resistance welded cladding, the specific inspection objectives (detection, discrimination, and sizing), and the inspection vendors integrated technology. The beltline inspections will be performed with a combination of eddy current and ultrasonic techniques. The specific methods will be defined when Yankee selects its inspection vendor. The selection process is under way with three vendors under consideration.

QUALIFICATION

The inspection methods for the beltline near surface examinations will be qualified to the intent of ASME Section XI, Appendix 8. Yankee currently is developing a qualification plate and practice blocks which will have the required number of near surface flaws in the proper distribution of size and location. The qualification plate will be carbon steel and clad with a similar material and process as used on the original vessel cladding or an acceptable alternative. The flaws will be implanted using the Hot Isostatic Pressure (HIP) process. The qualification block and practice blocks are being fabricated through the EPRI NDE Center. The NDE Center will be responsible for the security of the qualification block and the qualification process. They will administer the qualification tests jointly with Yankee.

INSPECTION BOUNDARIES

To satisfy the inspection objectives, Yankee will inspect a right circular cylinder with an upper bound of 9'-2" below the vessel flange parting line; a lower bound 7'-9" below the upper bound; an inside diameter equal to the Inside Diameter (ID) of the vessel; and an outside diameter of ID + 2.25". Yankee will use the upper and lower bound dimensions to define the beltline consistent with 10CFR50, Appendix G (1989), Paragraph II.F. The inside and outside diameters define the volume required to inspect both beltline welds and plate.

It is an objective of the inspection to examine 100% of the surface area to a depth of 1" as defined above. There may be conditions of access or geometry which prohibit a 100% examination. These prohibitions will be in large part due to specific design restrictions of the inspection equipment. Yankee will define these limitations, if any, when a complete inspection package is developed by the inspection vendor.

FLAW DEFINITION

ASME Section XI, 1986 Edition, will be used to define flaws. Specifically IWB-3510 will establish the acceptance criteria for surface and subsurface flaws. Flaws which are allowable per IWB-3510 will be acceptable for continued service. Yankee intends to use Table IWB-3510-3 specifically for eddy current examination criteria. A surface flaw, which is acceptable to IWB-3510-3 using eddy current inspection methods, will be considered acceptable for continued service without evaluation.

SCHEDULE

Yankee intends to make a preliminary determination of the scope and methods of inspection by February 1991. In early July 1991 a final determination of the scope of the 1992 outage will be made. Integrated testing of the inspection package is planned to begin in November 1991. Tooling is planned to be delivered to the site in April 1992 with inspection activities scheduled in early May 1992.

CONCLUSIONS

Yankee intends to develop, test, and deploy inspection equipment necessary to meet the inspection objectives. These objectives will be met when the beltline region is inspected and acceptable for continued operation based on criteria established in the 1986 Edition of the ASME B&PV Code, Section XI.

ATTACHMENT B

TEST REACTOR IRRADIATION

INTRODUCTION

Yankee met with the NRC on November 20, 1990 to discuss a proposed Test Reactor Irradiation Program. A submittal (Reference (d)) was made to the NRC describing the program and concurrence was received for the program from the NRC in Reference (e). A summary of the program is as follows. This is an update of the program described in Reference (d).

TEST OBJECTIVE

The objective of the Irradiation Test Program is to characterize the irradiation response of representative Yankee reactor vessel beltline plate materials and to remove uncertainties in the analysis of existing irradiation data. The uncertainties to be clarified are associated with the response of the beltline plate material to irradiation temperature (500°F versus 550°F), microstructure (coarse versus fine grain), and nickel content (high versus low) as described in Reference (b).

TEST MATERIAL SELECTION

Candidate plate materials for irradiation testing should have chemistry contents which closely match the contents of the Yankee beltline plates. The chemical elements of particular interest are copper and nickel. The plates to be matched are the Yankee lower and upper shell plates. Material (YA1 and YA2) has been found which approximates the copper and nickel content of the lower plate (Table I) and it is proposed that these materials be used for the irradiation testing corresponding to the lower plate. For the upper plate, two materials (YA8 and YA9 in Table II) have been identified which approximate its copper and nickel content. Yankee has possession of YA8 and has access to YA9. The preferred material is YA9 because it more closely approximates the other elements in the upper plate.

REFERENCE MATERIALS

Reference material will be used in the test program to verify irradiation results. The YA1 and YA2 materials were previously tested and results reported in a study published by R. Hawthorne in 1975 in ASTM STP-570 (pp 83-102). YA1 is "plate 2" of the study; YA2 is "plate 1" of the study. These plates were irradiated at several fluences in the range of interest and in both longitudinal and transverse orientations. These results can be used to benchmark results from the proposed test program. HSST-02 material is also available from the Heavy-Section-Steel-Technology (HSST) Program and will be used. HSST-02 plate material has been irradiated in power reactor (PWR) surveillance programs and these results can be used to determine flux rate or power versus test reactor effects on the irradiation test program.

ATTACHMENT B

TEST REACTOR IRRADIATION
(Continued)

HEAT TREATMENT

In order to duplicate the Yankee vessel steel coarse grain microstructure, the test materials must be heat treated. A heat treatment qualification program is being performed as shown in Figure 1. The qualification plates will be characterized in their as-received state, heat treated, and then tested for the desired microstructure. The process will be repeated until acceptable, repeatable results are obtained. The actual test plates will be heat treated upon completion of the qualification program. The acceptance criteria is based upon bringing the plate to an austenitizing temperature of 1,750°F to 1,800°F and cooling it to obtain a microstructure equivalent to the Yankee plate. The desired microstructure will be verified with photomicrographs.

IRRADIATION TEST MATRIX

The test matrix is shown in Figure 2. The first test will be the lower plate material at 500°F and 550°F at 3×10^{19} n/cm^2. The Charpy specimens will be oriented in the longitudinal direction to be consistent with previous testing. Two capsules will be irradiated with the following materials:

Capsule A (40 Specimens)
550°F Irradiation

Material	State	Type	Quantity
YA1	coarse	Charpy	12
YA1	coarse	tensile	2
YA1	fine	Charpy	12
YA1	fine	tensile	2
HSST-02	fine	Charpy	12
			40

Capsule B (50 Specimens)
500°F Irradiation

Material	State	Type	Quantity
YA1	coarse	Charpy	22
YA1	coarse	tensile	2
YA1	fine	Charpy	12
YA1	fine	tensile	2
HSST-02	fine	Charpy	12
			50

If the YA9 material, representative of the vessel upper shell, can be obtained in time, the first test matrix, Capsule B contents will be changed to contain:

Capsule B (Alternate)
500°F Irradiation

Material	State	Type	Quantity
YA1	coarse	Charpy	12
YA1	coarse	tensile	2
YA1	fine	Charpy	12
YA9	coarse	Charpy	12
YA9	coarse	tensile	2
HSST-02	fine	Charpy	10
			50

The proposed second irradiation test matrix is shown in Figure 3. It will simulate the upper plate and lower plate at 500°F to produce additional fluence data points. The capsule contents have not been fully established but are expected to be similar to the first irradiation. The second test reactor run will target fluence values of 1, 3, and 5×10^{19} n/cm^2 by removing specimens during the reactor run.

TEST REACTOR DOSIMETRY

At least two irradiations will be conducted in the University of Michigan's Ford test reactor. The flux at the core position to be used is estimated at 9×10^{12} n/cm^2/sec. The actual flux and neutron spectrum will be determined by irradiating a steel block containing dosimetry wires. Additionally, a dummy test will be conducted using the test capsules and test configuration to verify that capsule temperatures can be maintained at the two desired temperatures of 500°F and 550°F. Test temperatures are monitored throughout the irradiations using thermocouples. Materials Engineering Associates, Inc. (MEA) will encapsulate the test specimens and dosimetry and will conduct the irradiations. Laboratory analysis of dosimetry will be performed by EG&G. Babcock & Wilcox will determine the fluence by using their DOT 4.3 two-dimensional, neutron transport theory code with the following parameters:

- S8 Quadrature

- P3 Scattering

- ENDF/B4 Cross-Section Library

- BUGLE-80 Energy Group Structure

The following dosimetry wires will be used in each capsule containing test specimens:

DOSIMETRY WIRES	90% RESPONSE RANGE
Ni	2.1 - 7.6 Mev
Fe	2.5 - 7.8 Mev
Co/Al	Thermal
Ag/Al	Thermal
Nb	0.6 - 6.0 Mev
U-238	1.5 - 6.7 Mev

The U-238 will be encapsulated in either vanadium or stainless steel. It will then be placed in a gadolinium cover and finally an aluminum cover.

A calculation has been made of Displacements Per Atom (DPA) at the inside surface of the Yankee reactor vessel. The result was compared with a similar calculation at the center of a capsule containing specimens in an incore position of the Buffalo test reactor. The neutron spectrum at an incore position in the Ford test reactor should be similar to the Buffalo reactor. The thermal neutron contribution to the DPA at the inside surface of the Yankee reactor vessel was about 1.2% and 0.3% at the center of the test reactor capsule. The contributions to DPA from thermal neutrons is small for both the Yankee reactor vessel and the test reactor. Therefore, the irradiation test results from the tes_ reactor should be applicable to the Yankee reactor vessel plate material.

SCHEDULE

The first irradiation is scheduled to start in March 1991. To achieve the target fluence of 3E19 n/cm^2, a test duration of about 3 months is required. The second irradiation is expected to start in June 1991 and take four months.

CONCLUSIONS

The Irradiation Test Program is designed to accomplish several objectives.

1. Show the irradiation sensitivity of material microstructure and/or material heat treatment by irradiating the same test material with a fine and coarse grain microstructure.

2. Show the effects of irradiation temperature by irradiating the test material at 500°F and 550°F.

TEST REACTOR IRRADIATION
(Continued)

3. Develop the YA1 plate as a bounding material for the lower plate. The copper and nickel content of YA1 has a higher chemistry factor than the lower plate. It will be heat treated to develop equivalent microstructure to Yankee plate and will be irradiated at a temperature (500°F) equivalent to Yankee's irradiation temperature.

4. Provide test data for a high nickel plate to compare with the Yankee BR3 plate irradiation data to show the nickel effect. The BR3 data is for a Yankee plate of similar copper but lower nickel content than the test plate.

5. Confirm Yankee and BR3 surveillance results by testing YA9 material which is representative of the upper plate.

6. Produce at least two, and, if time permits three, fluence data points to define a transition temperature shift versus fluence trend for the YA1 and YA9 materials that would bound Yankee materials.

Figure 1

Heat Treatment Qualification

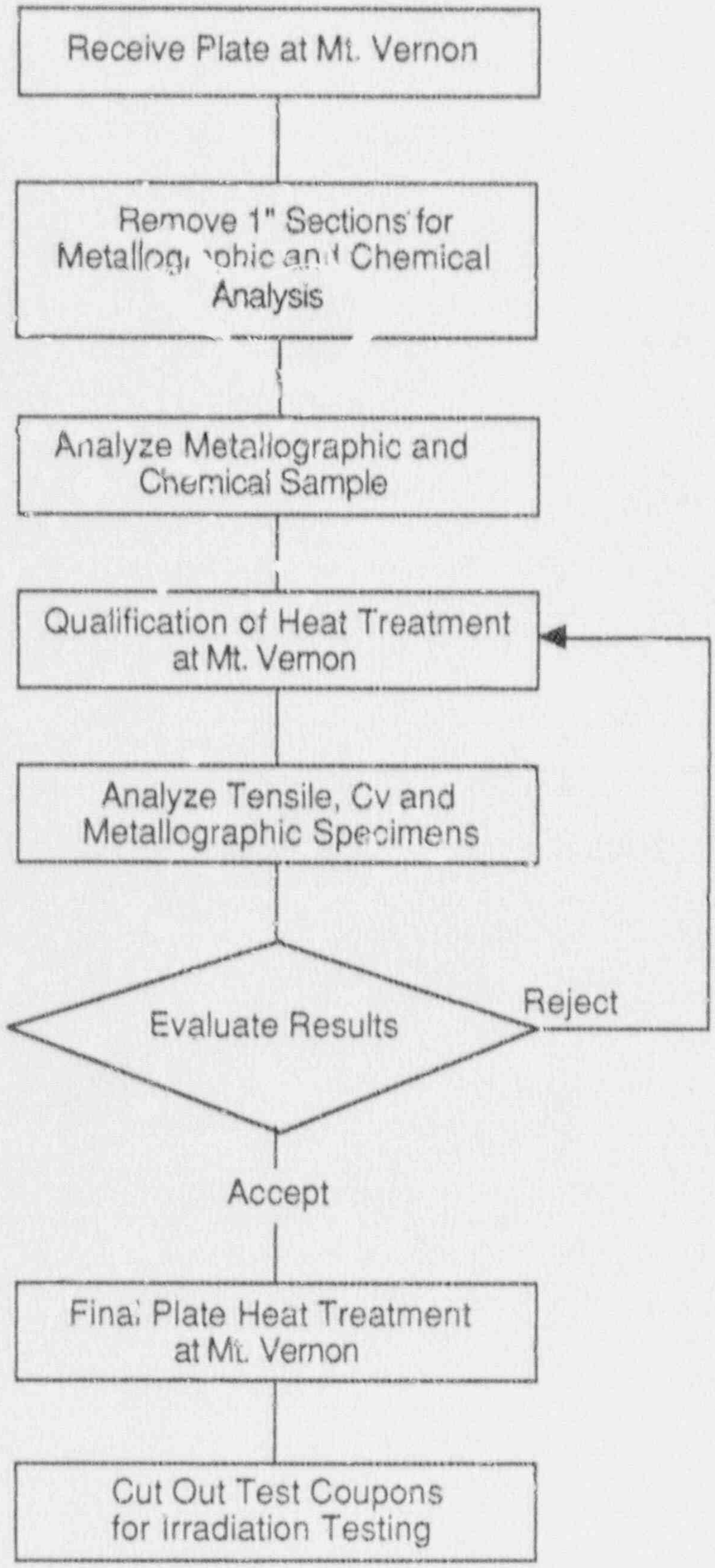
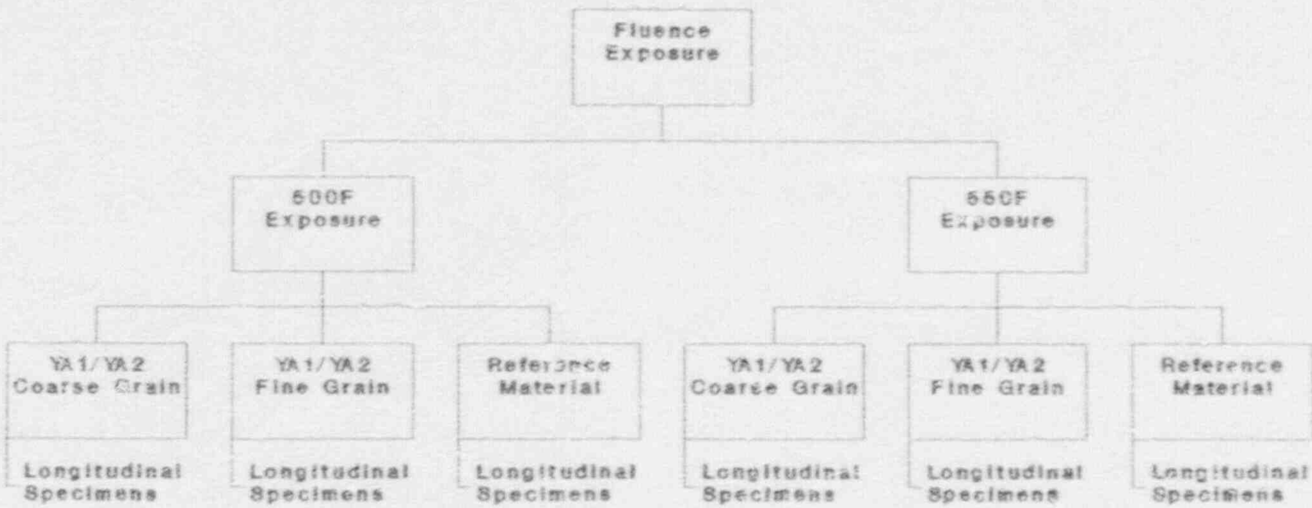


Figure 2

Irradiation Test Matrix for Lower Plate Material

A302-B Low Cu, Mod. Ni



Target Fluences

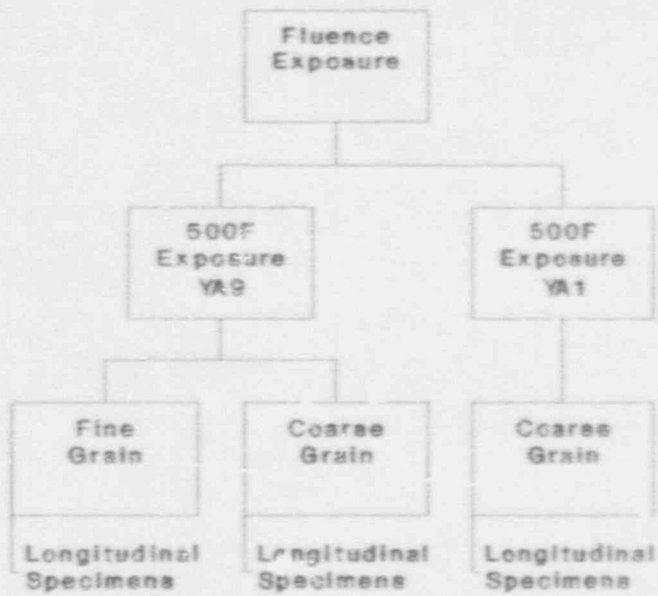
- *3.0 E19 n/cm-2 >1 MeV*
- *5.0 E19 n/cm-2 >1 MeV*

ATTACHMENT B

Figure 3

Second Irradiation Test Matrix for Upper Plate and Lower Plate Material

A302-B Low Cu, Low Ni



	1.0 E19 n/cm-2 >1 MeV
Target Fluences	*3.0 E19 n/cm-2 >1 MeV*
	5.0 E19 n/cm-2 >1 MeV

Three capsules will be used. The 5.0E19 n/cm2 capsule remains in the reactor for the entire run. The 3.0E19 n/cm2 capsule is pulled and replaced with another capsule for the 1.0E19 n.cm2 irradiations.

Attachment B

Table I

Lower Vessel Shell Materials

Chemistry

	Cu	Ni	C	Mn	Si	Mo	S	P	Cr	Al
YA1	.240	.620	.250	1.400	.230	.590	.011	.008	.110	.020
YA2	.170	.560	.230	1.290	.210	.570	.015	.009	.100	.027
Yankee Lower Plate	.200	.630	.190	1.180	.200	.480	.026	.016	.110	.020

Attachment B

Table II

Upper Vessel Shell Materials

Chemistry

	Cu	Ni	C	Mn	Si	Mo	S	P	Cr	Al
YA8	.140	.200	.210	1.150	.250	.600	.017	.015	.220	<0.01
YA9*	.240	.190	.170	1.280	.220	.500	.022	.026	.160	-
Yankee Upper Plate	.180	.210	.200	1.270	.210	.480	.028	.020	.060	-

* Not yet obtained.

ATTACHMENT C

WELD SAMPLING

INTRODUCTION

Yankee met with the NRC staff on November 20, 1990 to discuss preliminary plans for the Weld Sampling Program. The program has been expanded in response to NRC concerns expressed in the meeting. The following is a summary of the proposed Weld Sampling Program:

OBJECTIVE

The objective of the Weld Sampling Program is to remove sufficient material to determine the elemental chemistry, primarily copper and nickel, of the beltline circumferential and intersecting axial welds. The sample depressions will be sized to maintain the code-required structural integrity of the vessel.

SAMPLE LOCATIONS

Weld material samples are planned to be taken from the reactor vessel circumferential and two axial welds located in the core beltline region. The proposed number and locations of the samples to be taken are shown on Figure 1. Three samples will be taken at each of the two circumferential-axial weld intersecting points, with an additional sample taken from each axial weld. Reviews are being made to assure that the proposed sample locations are not in areas of weld repair.

SAMPLE SIZE

The welding and cladding processes used on the vessel welds resulted in a weld cross section similar to Figure 2, which shows that the bulk weld metal will be reached at a depth of approximately 1/2" from the surface.

The proposed material samples to be removed will be approximately 0.75" x 0.75" x 0.75" deep. The cube shaped samples will be removed and sectioned to determine chemistry as a function of depth.

Following sample removal, the depressions will be prepared to remove all edges, leaving rounded bottom cylindrical depression with a diameter less than 2" and a depth not exceeding 1". The surface edges shall also be chamfered to approximately a 1/8" radius.

DESIGN CONSIDERATION

The proposed sample size, shape, number, and locations are based on preliminary assessments of code limitations and the ability to obtain them. Formal calculations, including localized stress analysis of the sample sites, are being prepared to assure code acceptability of the proposed sample locations and depressions using the flaw evaluation rules of ASME Section XI, IWB-3610 (d), which applies the criteria of Section III of NB-3213.10 and NB-3221.2.

SAMPLING METHOD

The method of sample removal is expected to be by Electric Discharge Machining (EDM). Other possible sample geometries and methods of removal, however, are still under consideration.

ACCESS CONSIDERATIONS

All of the proposed sample locations, except one, are located behind the thermal shield. The shield, shown on Figures 3 and 4, is made of 3" thick stainless steel plate. The radial gap between the thermal shield and reactor vessel wall is nominally 2", with some areas expected to be approximately 1-1/2".

While it would be preferred to take samples from behind the thermal shield, it is anticipated that access holes will need to be machined through the thermal shield. Two large access holes at the weld intersection points and one smaller access hole for the upper axial weld are being considered.

The large holes would be sized to allow access to all three sample sites. The access holes will then be repaired. The repair plugs will be installed using mechanical methods without welding. The plugs will be positively secured, with bolting designed to preclude loosening during anticipated operating conditions.

The thermal shield access hole plugs and hardware will be fabricated of stainless steel.

SCHEDULE

Code calculations will be completed in January 1991. A preliminary determination of the scope and method of sampling is scheduled for the end of March 1991. Demonstration testing is scheduled for early 1992, and actual sampling will take place during the spring 1992 refueling outage. The chemical content of the samples will be determined within days of the sample removal.

Attachment C

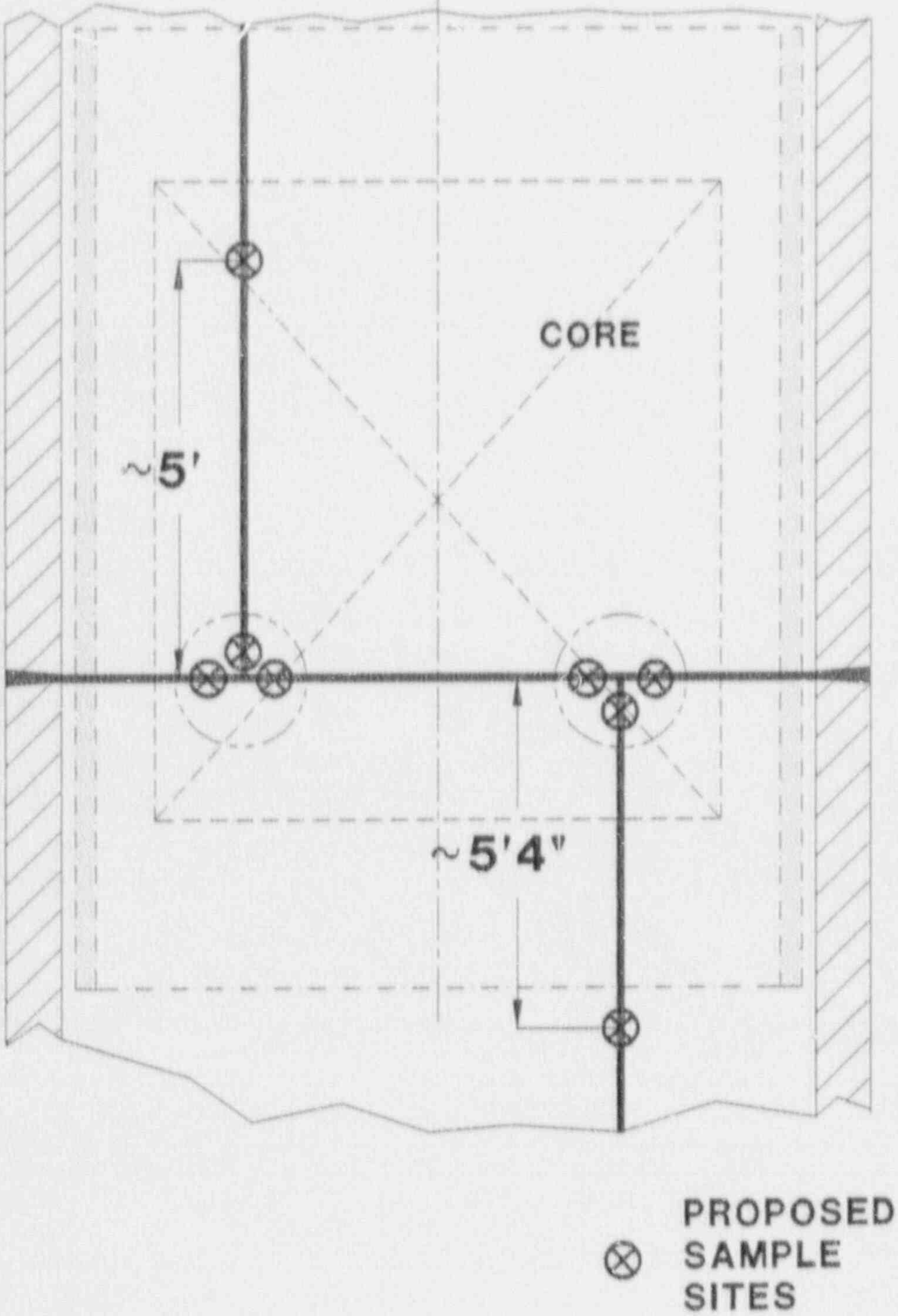


Figure 1

Attachment C

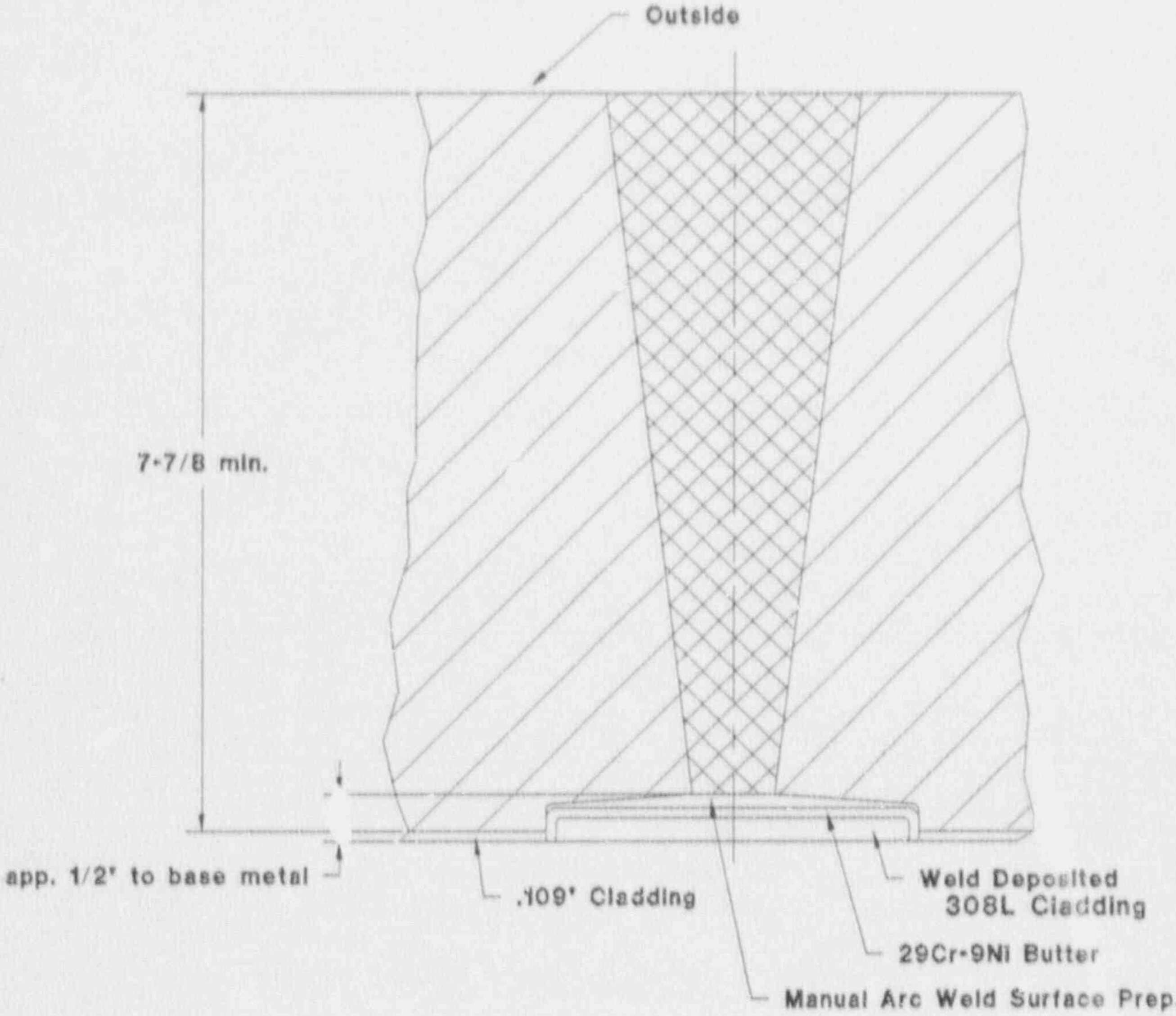


Figure 2

Attachment C

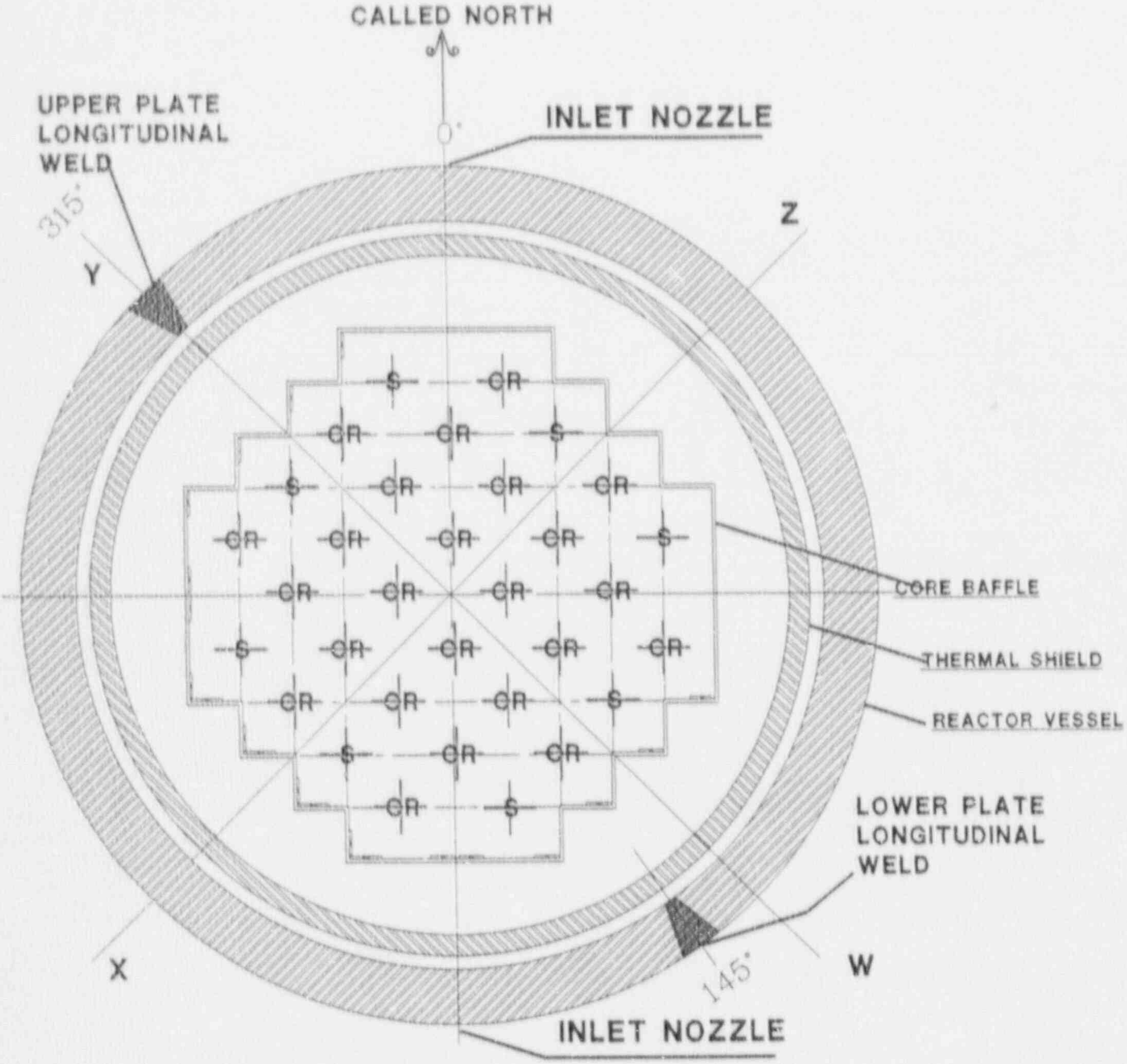


Figure 3

Attachment C

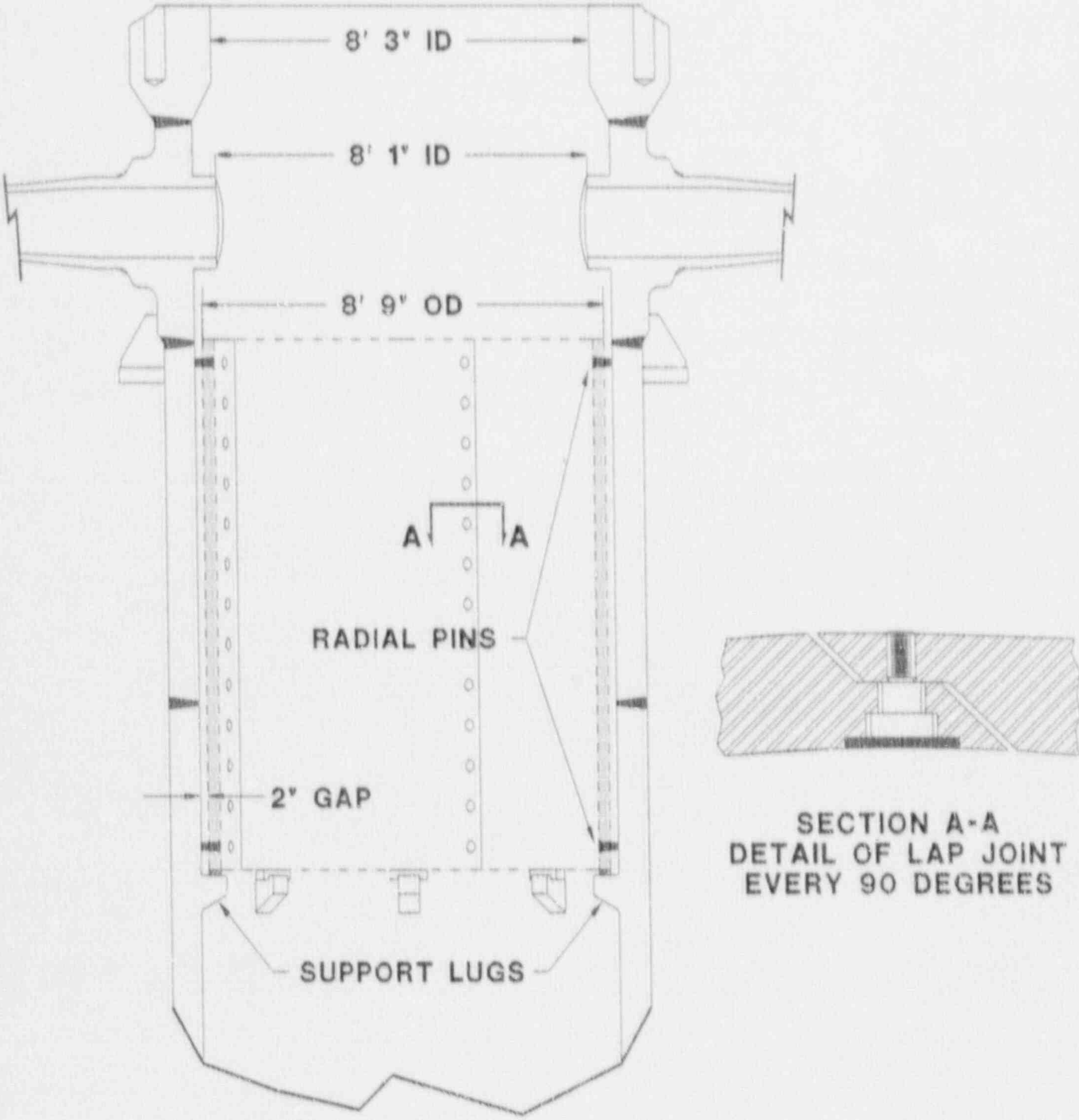


Figure 4

ATTACHMENT D

SURVEILLANCE CAPSULE

INTRODUCTION

The NRC Safety Assessment (Reference (b)) requires the installation of surveillance capsules in accelerated irradiation positions in the Yankee reactor pressure vessel. In a meeting with the NRC staff on September 18, 1990, the intent of the requirement was clarified. The installation of the capsules was not to be required until the outage after the 1992 outage because the capsules were to contain weld metal specimens which cannot be obtained until the weld chemistry is determined at the 1992 outage.

OBJECTIVE

The objective of re-establishing a surveillance program in the Yankee reactor is to show irradiation effects on material similar to Yankee beltline plates and weld under actual Yankee irradiation conditions.

MATERIALS

The materials to be placed in the capsules will be representative of the beltline materials. YA1 or YA2 from test reactor irradiation program (Attachment B) will be used to correspond to the lower plate, and YA9 will be used to correspond to the upper plate. The selection of the weld metal will depend upon the chemistry content determined by weld sampling (Attachment C). Once the chemistry is known, similar material will be found or manufactured for placement in the capsules.

LOCATION

The locations of the capsules are being chosen based upon available space and the amount of accelerated irradiation the capsules will experience. Locations between the core barrel and the thermal shield are being considered. The capsules would be attached to either the barrel or thermal shield. The neutron flux associated with these locations is being calculated. A lead factor of 4 to 8 is desired.

CAPSULE DESIGN

The capsule design and installation details will be based upon the locations selected. The configuration, contents, and dosimetry will be determined once the location and weld chemistry are established.

SCHEDULE

A list of candidate locations and attributes will be complete by the end of March 1991. Selection of locations will be made by the end of April 1991. The tooling and hardware needed for installation will be designed and procured by the end of 1991. Our intent is to modify the internals for capsule installation during the 1992 outage and install capsules during the following outage.